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# Neutron Spectral Considerations Affecting Projected Estimates of Radiation Embrittlement of the Army SM-1A Reactor Pressure Vessel

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## ABSTRACT

The pressure vessel of the Army SM-1A reactor is located close to the active core in such a manner that the neutron exposure is relatively high; consequently, the pressure vessel steel undergoes a relatively rapid rise in the ductile-brittle transition temperature. The maximum permissible ΔNDT for the SM-1A is established by the Army as 340°F. Since it is physically impossible to irradiate surveillance test specimens at the SM-1A vessel wall, only the neutron flux was measured at the wall, and representative test specimens were irradiated in a test reactor, the Low Intensity Test Reactor (LITR). In translating the ΔNDT versus neutron exposure data from the LITR to the case of the SM-1A reactor vessel wall, the neutron spectra of the two reactors were used to adjust both the SM-1A reactor vessel flux and the LITR exposure values in terms of $n/cm^2 > 1.0$, 0.5, and 0.183 Mev. Since the distribution of neutrons by energy groups was different within each reactor at the specific location of interest, that is, the vessel wall of the SM-1A and an in-core location of the LITR, the damaging potential of the SM-1A reactor spectrum location was related to that of the LITR.

With damage equivalence established between the two reactors, a critical neutron exposure ($n/cm^2 > 0.5$ Mev) may be projected for producing the maximum ΔNDT on the SM-1A reactor vessel wall. By relating this critical exposure to SM-1A reactor operations, a critical power output level of 67 Mw-yr was established. The same Mw-yr critical power output levels will be obtained if the neutron exposure is reported as $n/cm^2 > 1$ Mev or $n/cm^2 > 0.183$ Mev. If the reactor spectra were assumed to be fission spectra, however, the Mw-yr critical power output limit for neutrons > 1 Mev, for instance, would be 49.5. Thus, by considering all of the neutrons actually present in a reactor spectrum rather than by assuming a fixed distribution, a more precise calculation of the neutron exposure required to effect a NDT increase can be determined, and less conservatism need be applied in projecting safe operating limits.

## PROBLEM STATUS

This completes one phase of the problem; other phases of this research effort are continuing.

## AUTHORIZATION

NRL Problem M01-14
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# NEUTRON SPECTRAL CONSIDERATIONS AFFECTING PROJECTED ESTIMATES OF RADIATION EMBRITTLEMENT OF THE ARMY SM-1A REACTOR PRESSURE VESSEL

## INTRODUCTION

The Army SM-1A reactor is a pressurized, light water moderated plant of 20.2 MW (thermal) capacity located at Fort Greely, Alaska. The plant is used by the Army to supply heat and electricity to the post. The SM-1A was designed and built by Alco Products, Inc., according to the philosophy that field plants such as this should be relatively compact for ease of transporting to remote stations. As a result, the SM-1A has a pressure vessel of small diameter, located relatively close to the nuclear core although separated from it by two thermal shields (Fig. 1).

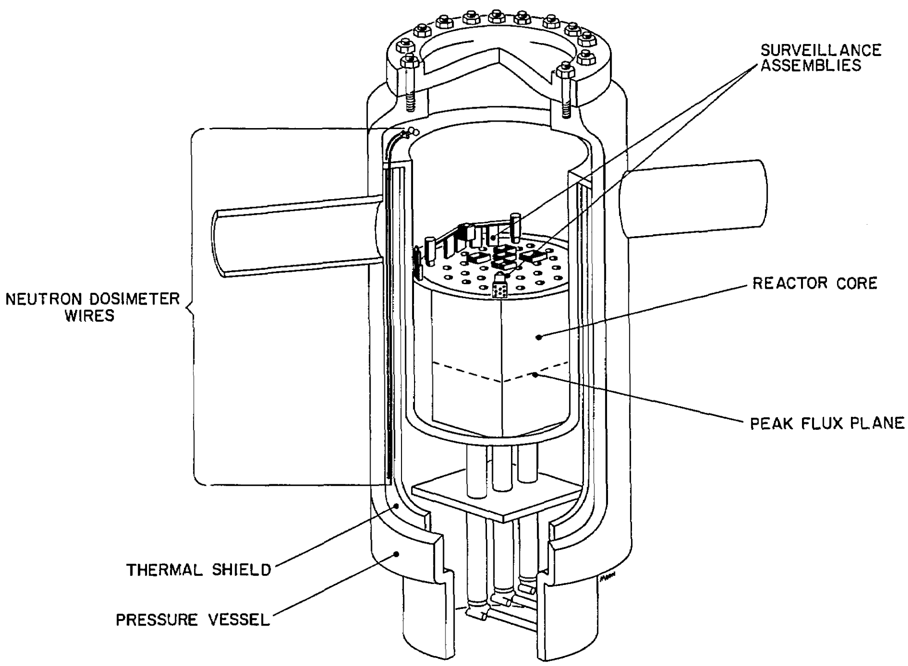


Fig. 1 - Schematic view of the SM-1A reactor showing positions of encapsulated surveillance assemblies and neutron dosimeter wires with reference to the core and to the critical pressure vessel regions

The steel for the pressure vessel, A350-LF1 (Modified), was selected partly because of its low initial nil-ductility transition (NDT) temperature; that is, the temperature at which the steel undergoes a change from brittle (low energy) to ductile (high energy) fracture characteristics. More specifically, the NDT is defined as that temperature at or below which a small flaw residing in a zone of yield point loading will initiate rapid brittle failure. Carbon and low-alloy steels, however, undergo a rise in this transition temperature (indicated by ΔNDT) as a result of bombardment by high energy neutrons. With continued neutron exposure during reactor operation, the NDT of the vessel steel could rise to a point near or even above the operating temperature of the vessel. As demonstrated by the analysis of failure conditions of several non-nuclear components (1), it is hazardous to operate components near the NDT temperature when the critical conditions of temperature, stress, and flaw size may exist. With such conditions, a sudden operational irregularity or any detrimental change in one of the failure conditions, especially stress or flaw size, may result in a catastrophic brittle failure of the vessel.

It was recognized that the nearness of the pressure vessel to the core of the SM-1A presented the possibility of more rapid accumulation of neutron exposure at the vessel than normally would be encountered in power reactor operations. Consequently, the Army requested that the neutron flux at the vessel wall be measured, and that Charpy V-notch test specimens of a fabrication test plate and ring forging representative of the vessel be placed in the reactor in surveillance locations to monitor the upward shift in the NDT. Since the neutron exposure in the SM-1A reactor at the accessible surveillance locations would be at a low rate, the Army further requested that additional specimens of this test plate and ring forging be irradiated under accelerated conditions in a test reactor to obtain data which permit extrapolation of properties of the pressure vessel after specific periods of operation.

In the unirradiated condition, the NDT of the test plate was -40°F and that of the forging -70°F. In the interest of conservatism the Army has assumed the SM-1A vessel forging NDT to be at the higher level of -40°F. Evaluation of the results of Charpy V-notch specimen tests following irradiation under accelerated conditions revealed that the A350-LF1 (Modified) steel ring forging, which was typical of the forged sections of the vessel wall, developed a ΔNDT of 340°F with a neutron exposure of $2.0 \times 10^{19}$ n/cm$^2$ > 1 Mev. Coincidentally, the maximum allowable ΔNDT for the SM-1A vessel steel has been placed by the Army at 340°F. Thus, the limiting neutron exposure to the vessel steel should be $2.0 \times 10^{19}$ n/cm$^2$ > 1 Mev. This neutron exposure was determined by assuming a fission neutron spectral shape for the flux spectrum in the Low Intensity Test Reactor (LITR) where the experiment was performed, for the iron dosimeter ($Fe^{54}$(n,p)), using an activation cross section of 68 mb and extrapolation along the fission spectrum to determine the number of neutrons above 1 Mev. The neutron flux intensity at the SM-1A vessel wall was also determined using the same techniques and assumptions. With this information, it thus became a relatively simple matter to convert the flux at the vessel wall into a megawatt-year limit for the reactor vessel to reach the 340°F ΔNDT.

A simple analysis as presented above is not possible, however, since neither the LITR spectrum nor the SM-1A vessel wall spectrum have the exact shape of a fission neutron spectrum. This report describes the neutron spectra of these and other pertinent irradiation locations, the technique used to calculate them, and the neutron exposures adjusted by the application of spectral data and correlated experimental ΔNDT data to yield more accurate megawatt-year limit estimates for reaching a critical NDT on the SM-1A reactor vessel.

## EXPERIMENTAL DATA

### Neutron Flux at the Pressure Vessel Wall

Stainless steel tubes containing iron, nickel, and cobalt dosimeter wires were placed alongside the SM-1A pressure vessel wall just prior to startup of the plant (2). The iron monitor tube was exposed during the first two calendar months of low-power tests for a total of 1023 hours at an average power level of 8.65 Mw. These exposure data have been derived from a thorough evaluation of the steam power output history during that period and reflect an adjustment of the exposure data reported (2). Subsequent recalculations of the monitor activity for derivation of the neutron flux at the vessel wall based upon this new information shows that the peak flux at the vessel wall is $1.11 \times 10^{11}$ n/cm$^2$-sec $>1$ Mev, assuming a fission spectrum and a fission-averaged iron activation cross section of 68 mb. The neutron flux gradient along the vessel wall is shown in Fig. 2.

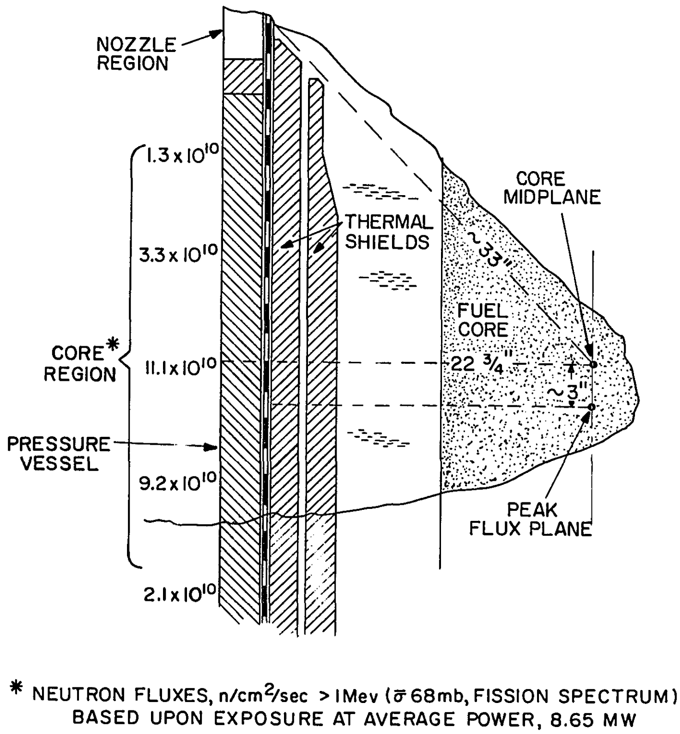


Fig. 2 - Schematic view of the critical pressure vessel region of the SM-1A reactor and the instantaneous neutron flux values measured using the iron dosimeter wires along the vessel wall. The flux was based upon activation of 304 stainless steel tubing during reactor operations at an average power level of 8.65 Mw.

Extrapolation of the $1.11 \times 10^{11}$ flux measured at 8.65 Mw to the full power of the SM-1A yields a peak vessel-wall flux ($>1$ Mev) of $2.59 \times 10^{11}$ at 20.2 Mw. Thus, the total exposure for 1 megawatt-year (Mw-yr) is $4.04 \times 10^{17}$ n/cm$^2$ $>1$ Mev, and one full power-full operating year (20.2 Mw-yr) is $8.17 \times 10^{18}$ n/cm$^2$ $>1$ Mev. Note that all these values are based upon an assumed fission spectrum and a fission-averaged iron activation cross section of 68 mb.

## Vessel Steel ΔNDT Data

Steel from the forgings which were used in the construction of the SM-1A pressure vessel were not available for test specimen preparation. However, a 3-5/8-in.-thick fabrication test plate (hereafter referred to as plate) and a 3-5/8-in.-thick duplicate ring forging (hereafter referred to as forging) of the SM-1A steel, with a composition and heat treatment history which duplicated the construction forgings, were used for the irradiation studies. The latter is considered more representative of the vessel than the plate, though both exhibit very similar mechanical properties. This steel was modified from ASTM Type A350-LF1 steel primarily by the addition of nickel (1.7% Ni).

Initial notch ductility properties of the steels were determined by testing both Charpy-V and drop weight test specimens. The plate NDT was -40°F and forging NDT was -70°F. The 30-ft-lb Charpy-V point was slightly lower in both cases (Table 1). A conservative NDT value of -40°F has been selected by the Army as the NDT representative of the SM-1A reactor vessel.

TABLE 1
Charpy-V Transition Temperature Behavior of A350-LF1 (Mod.) Fabrication Test Plate and Duplicate Ring Forging Representing the SM-1A Reactor Pressure Vessel

| Material | Reactor Facility | Irradiation Temperature (°F) | Neutron Exposure ($n/cm^2$ > 1 Mev) | Charpy-V 30-ft-lb Transition Temp. (°F) | | | Full Shear Energy Absorption (ft-lb) | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | Initial | Final | ΔT | Initial | Final | Δft-lb |
| A350-LF1 (Mod.) | LITR: | | | | | | | | |
| Plate | C-18 | 430 | $2.0\times10^{19}$ | -45 | 285 | 330 | 122 | 68 | 54 |
| Forging | C-18 | 430 | $2.0\times10^{19}$ | -80 | 260 | 340 | 129 | 70 | 59 |
| Plate | C-55 | 430 | $2.8\times10^{19}$ | -45 | 370 | 415 | 122 | 57 | 65 |
| Forging | C-18 | 430 | $2.8\times10^{19}$ | -80 | 300 | 380 | 129 | 71 | 58 |
| Plate | C-18 | 430 | $3.1\times10^{19}$ | -45 | 350 | 395 | 122 | 82 | 40 |
| Plate | C-55 | 430 | $3.1\times10^{19}$ | -45 | 395 | 440 | 122 | 55 | 67 |
| Plate | SM-1A above core | 445-475 | $2.6\times10^{18}$ | -45 | 35 | 80 | 122 | 96* | ≥26 |

*90% shear (single test point).

Low-temperature (240°F) as well as vessel-operating-temperature (430°F) accelerated irradiations of Charpy-V specimens of both the SM-1A plate and forging steels have been conducted in core positions C-18 and C-55 of the LITR at the Oak Ridge National Laboratory. The lower temperature irradiations were useful in the determination of the general trend behavior of the steels, but for the purposes of this report, only the irradiations conducted at 430°F will be considered. These data have been previously reported (3) and are summarized in Table 1.

The data in Table 1 reveal that one irradiation experiment in position C-18 of the LITR resulted in a ΔNDT for the forging steel of 340°F, considered to be the maximum allowable increase for the SM-1A vessel. In Fig. 3 it is noted that the remainder of the LITR data as well as the SM-1A reactor surveillance data point tend to confirm the location of this data point along the trend line for steel behavior. Therefore, this point will be used as the primary reference for the long term exposure analysis of the SM-1A pressure vessel.

## NEUTRON SPECTRAL CONSIDERATIONS

Neutron dosimetry for irradiation experiments is based upon the activation of neutron dosimeter wires at relatively high energy levels. The most used dosimeter, iron, is activated primarily by neutrons having energies between 2 and 8 Mev. And, since a reactor neutron spectrum has relatively few neutrons with energies above ~8 Mev, it is

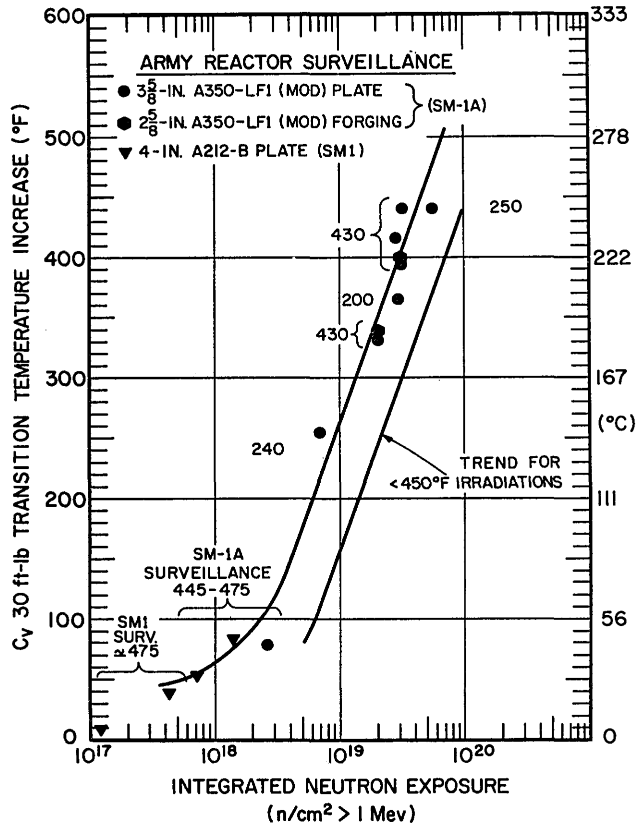


Fig. 3 - Charpy-V 30-ft-lb transition temperature increase versus integrated neutron exposure for SM-1A pressure vessel steel referenced to <450°F trend band irradiations. Data points near the low exposure end of the trend band are from SM-1A surveillance program. Points indicating material behavior at higher exposures are from accelerated irradiations in test reactors. Data for A212-B steel from SM-1 and SM-1A surveillance programs are shown for reference.

reasonable to say that most of the iron dosimeter activation is caused by neutrons in a "window" varying from about 2 to 8 Mev. The extent of activation is determined, and thus is directly related to both the shape and the intensity of the spectrum. In order to obtain a value for neutrons above 1 Mev, it is necessary to then extrapolate along some neutron spectral shape from an effective threshold of about 3 or 4 Mev down to 1 Mev. Commonly, a fission neutron spectral shape is assumed (Fig. 4), and the flux intensity above 1 Mev is determined therefrom. If the irradiation spectrum deviates from a fission neutron spectral shape, the resultant flux value will be in error to a degree. The magnitude of this error will be in relation to the magnitude of the deviation of the actual spectrum from that of the assumed fission spectrum.

In order to most accurately determine the neutron exposure for a particular reactor location, it is necessary to calculate the neutron spectrum for that location and to redetermine the dosimeter activation cross section for that spectrum. Recent advances in theoretical reactor physics and computer techniques now make it possible to reduce the

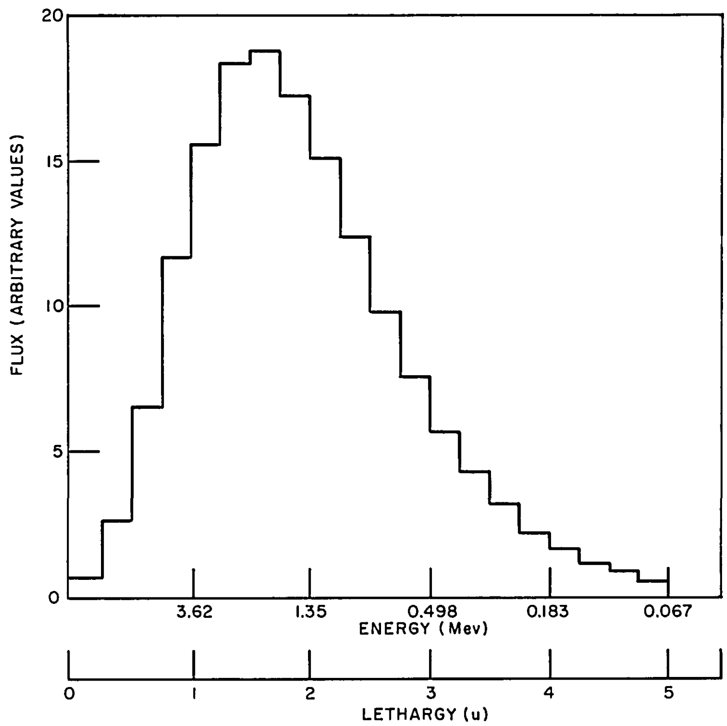


Fig. 4 - Graphical representation of the Watt fission spectrum plotted by arbitrary flux values in terms of $\phi(u)$ versus 0.25 lethargy ($u$) units

problem of calculating neutron spectra and revised cross section values to manageable proportions.

## Dahl-Yoshikawa Spectral Correction Approach

Dahl and Yoshikawa of the Battelle-Northwest Laboratory have studied the problem of neutron spectral differences in nuclear reactors and have evolved a technique whereby damage in one reactor can be correlated to damage in another reactor (4,5). The technique involves the use of transport theory, Sn reactor physics codes for both one-dimensional (6), and two-dimensional configurations (7). Studies presently going on at Battelle-Northwest have indicated that the Sn transport theory method of calculating neutron spectra will yield spectral structural detail as fine or even finer than that afforded by the method used in the P1MG and P3MG type codes.

Having a theoretical spectrum for a particular reactor, Dahl and Yoshikawa are then able to redetermine average dosimeter activation cross sections for neutrons whose energies are greater than selected "thresholds," or, more correctly, certain lower energy limits within that spectrum. These new cross sections permit a much more accurate determination of the neutron populations above selected lower energy limits in a given spectrum. Accordingly, radiation damage effects arising from exposures in diverse reactor environments may then be better interrelated, since the variations in the neutron spectra cannot invalidate the neutron dosimetry values.

Dahl and Yoshikawa have concluded that damage from diverse reactor exposures can be best interrelated if neutron exposures are presented in terms of a lower energy limit of 0.5 Mev (5). The basis for this conclusion is described briefly.

Gross atomic displacement production appears to be a logical fundamental process to employ as a basis for radiation-damage correlations, since mechanical property changes are initiated by displacements. They assume, then, that whatever the defect structure may be which influences a chosen property, accumulation of these defects will bear the same relationship to displacement production if all environmental factors other than neutron spectra are constant.

It was found convenient to determine a proportionality constant $K$ between the displacement rate for all of the neutrons in a given spectrum $D$ and the integrated flux $\phi$ of neutrons in that spectrum whose energies were above arbitrarily selected lower energy limits $E_L$ according to the equation

$$K_i = \frac{\int_0^\infty \phi(E)\ \Sigma_s(E)\ N(E_n)\ dE}{\int_{E_L}^\infty \phi(E)\ dE} = \left(\frac{D}{\phi_{E_L}}\right)_i \tag{1}$$

where $K_i$ is the proportionality constant for the $i$th reactor, $\phi(E)$ is the differential neutron spectrum in terms of $cm^{-2}$ $sec^{-1}$, $\Sigma_s$ is the differential macroscopic scattering cross section in terms of $cm^{-1}$, and $N(E_n)$ is the number of displacements caused by one neutron $n$ of energy $E$.

Note most carefully that the displacement rate accounts for every neutron within the neutron spectrum. In determining $K$ factors, only the lower energy limit $E_L$ is varied until the same or nearly the same $K$ is obtained regardless of the spectrum being considered. In practice, the damage model of Kinchin and Pease (8) was used to calculate the displacements in iron for a Watt fission spectrum, a spectrum in a light water moderated test reactor and in a well moderated graphite reactor spectrum. The three curves of displacement versus lower energy limits converge near 0.4 Mev and thus determine a meaningful lower energy limit for accounting of neutrons in diverse spectra for iron.

As a test of the sensitivity of the method to the damage model, Dahl and Yoshikawa further used the model of Rossin (9), which places more emphasis upon the higher energy neutrons than does the Kinchin and Pease model. While the Kinchin and Pease model yields a lower energy limit of 0.41 Mev, the Rossin model yields a lower energy limit of 0.439 Mev. Therefore, this technique for determining a lower energy limit for assinment of an encompassing effective "threshold" for radiation damage in iron appears to be insensitive to the most uncertain quantity in the calculation — the damage model (5).

The above technique was extended to include spectra from unmoderated fast reactors and heavy water moderated reactors. Again, the displacements per unit of flux were found to be most closely correlated if a lower energy limit of 0.5 Mev (chosen for convenience over 0.4 Mev because of the computer code output characteristics) were employed for the accounting of neutrons. When lower energy limits of 1 Mev or 0.183 Mev were employed instead of 0.5 Mev, variations in the calculated displacements per unit of flux for the diverse spectra became significantly greater.

As a critical test of this thesis, Dahl and Yoshikawa plotted transition temperature increase data from NRL experiments conducted in reactors of significantly differing moderating media versus exposures calculated for neutrons above three different $E_L$ values. Normalization of the data points into a single body of data was achieved using an $E_L$ of 0.5 Mev; however, a value of 1 Mev for $E_L$ did not achieve normalization. Further, extension of the technique using an $E_L$ of 0.183 Mev also did not significantly improve the data normalization. Thus the data normalization as well as the theoretical studies carried out by Dahl and Yoshikawa demonstrate that when all the neutrons actually present in diverse spectra are considered, damage can be best interrelated between the reactors by reporting neutrons whose energies are above a lower energy limit of 0.5 Mev.

## SM-1A Reactor Spectra

The neutron spectrum at the flux monitor tube location alongside the pressure vessel wall of the SM-1A reactor is shown in Fig. 5 as calculated by Program S (6) (one dimension). The calculated arbitrary unit fluxes per energy group which define this spectrum are shown in Table 2. It should be noted that this monitor tube location was 1/4 in. inside the stainless steel cladding of the vessel wall; which in turn provides a cover of approximately 1/4-in. thickness over the vessel steel. The spectrum at the pressure vessel edge (inside the cladding) differs from that at the monitor tube location somewhat in intensity, but only very slightly in shape.

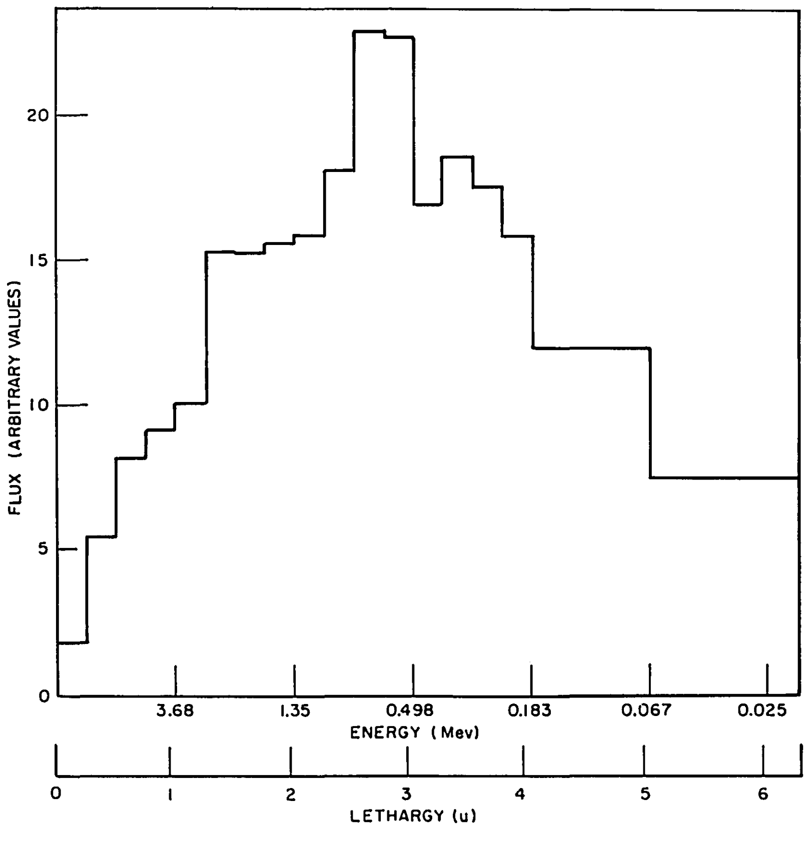


Fig. 5 - Graphical representation of the neutron spectrum at the neutron flux monitor position of the SM-1A reactor centered 1/4 inch closer to the core than the stainless steel vessel cladding

As shown in Fig. 1, Charpy-V specimens were placed in a surveillance location above and near the outer periphery of the core. The neutron spectrum of this above-core location is shown in Fig. 6 as calculated by Program S (6). The calculated arbitrary unit fluxes per energy group for this spectrum are given in Table 2.

TABLE 2
Summary of Calculated Fluxes per Lethargy Group and Lower Energy Limit for LITR Core Lattice Facilities, SM-1A Reactor Pressure Vessel Wall, and Above-Core Surveillance Position

| Lethargy ($U_L$) | Energy $E_L$ (ev)* | Calculated Arbitrary Unit Flux | | | |
|---|---|---|---|---|---|
| | | LITR Core Position | | SM-1A | |
| | | C-18 | C-55 | Above-Core Location | Pressure Vessel Wall |
| 0.25 | $7.79 \times 10^6$ | 8.72 | 7.07 | | 1.76 |
| 0.50 | $6.07 \times 10^6$ | 24.2 | 21.4 | 6.86 | 5.37 |
| 0.75 | $4.72 \times 10^6$ | 54.1 | 46.6 | | 8.20 |
| 1.00 | $3.68 \times 10^6$ | 87.4 | 69.0 | 16.07 | 9.15 |
| 1.25 | $2.87 \times 10^6$ | 136.0 | 88.3 | | 10.57 |
| 1.50 | $2.23 \times 10^6$ | 195.0 | 162.0 | 25.79 | 15.27 |
| 1.75 | $1.74 \times 10^6$ | 205.0 | 215.0 | | 15.14 |
| 2.00 | $1.35 \times 10^6$ | 236.0 | 239.0 | | 15.53 |
| 2.25 | $1.05 \times 10^6$ | 226.0 | 212.0 | 29.46 | 15.80 |
| 2.50 | $8.21 \times 10^5$ | 217.0 | 200.0 | | 18.06 |
| 2.75 | $6.39 \times 10^5$ | 224.0 | 199.0 | | 22.87 |
| 3.00 | $4.98 \times 10^5$ | 186.0 | 168.0 | 14.55 | 22.66 |
| 3.25 | $3.88 \times 10^5$ | 146.0 | 144.0 | | 16.87 |
| 3.50 | $3.02 \times 10^5$ | 159.0 | 148.0 | | 18.58 |
| 3.75 | $2.35 \times 10^5$ | 138.0 | 128.0 | | 17.52 |
| 4.00 | $1.83 \times 10^5$ | 119.0 | 111.0 | 8.52 | 15.59 |
| 5.00 | $6.74 \times 10^4$ | — | — | 3.24 | 47.67 |
| 9.00 | $1.23 \times 10^3$ | — | — | 5.66 | 121.19 |
| 13.00 | 22.6 | — | — | 9.51 | 176.60 |
| 17.00 | 0.414 | 3590 | 3460 | 2.60 | 198.84 |
| ∞ | 0 | 820 | 1000 | 890.16 | 135.81 |

*$E_L$ is the lower energy limit of the group.

## Low Intensity Test Reactor (LITR) Spectra

The accelerated irradiation experiments of SM-1A vessel steel were performed in core lattice positions C-18 and C-55 of the LITR (Fig. 7). The neutron spectra of the C-18 and C-55 positions are shown in Figs. 8 and 9 respectively; the computer code used for these calculations was Program 2DXY (7) (two dimensions). The calculated arbitrary unit fluxes per energy group for these spectra are given in Table 2.

## Fission-Averaged Activation Cross Sections

Prior to the adjustment of fluxes to conform to the neutron spectral characteristics of each irradiation location, it is necessary to reduce the monitor activity to a flux value. Commonly, this flux is stated in terms of n/cm$^2$ > 1 Mev, assuming a fission spectrum. The most important physics constant in this calculation is the cross section for activation of the flux monitor material. Depending upon the value used for this constant, the resultant fluxes can vary by a factor of almost two.

Accelerator studies have been carried out on iron to determine the activation as a function of energy for the reaction $Fe^{54}$(n,p)$Mn^{54}$. This reaction has been employed as

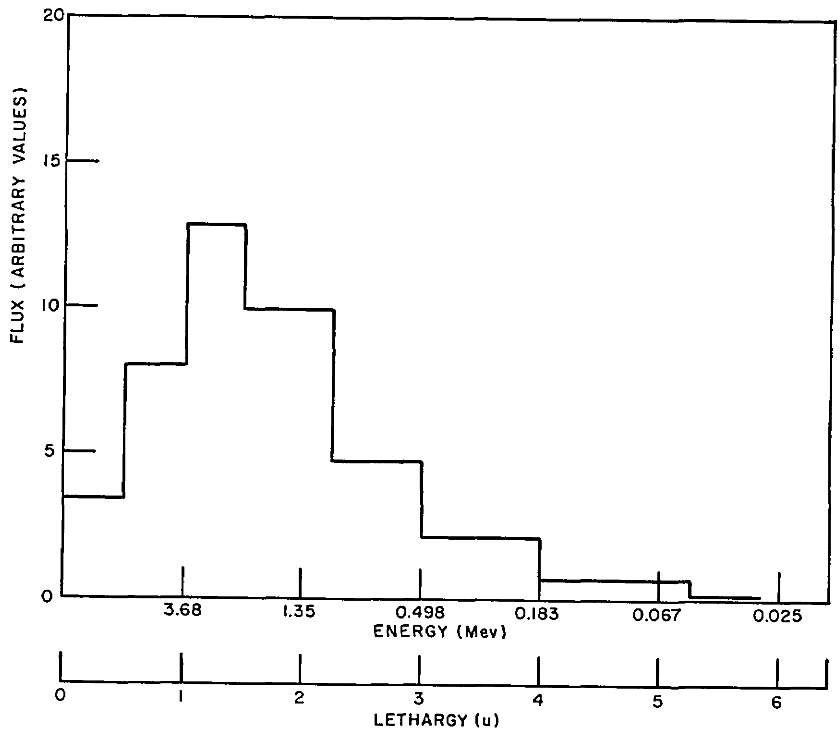


Fig. 6 - Graphical representation of the neutron spectrum at the Charpy specimen V-notches in the lowest specimen section of the surveillance capsules located above and at the edge of the core of the SM-1A reactor

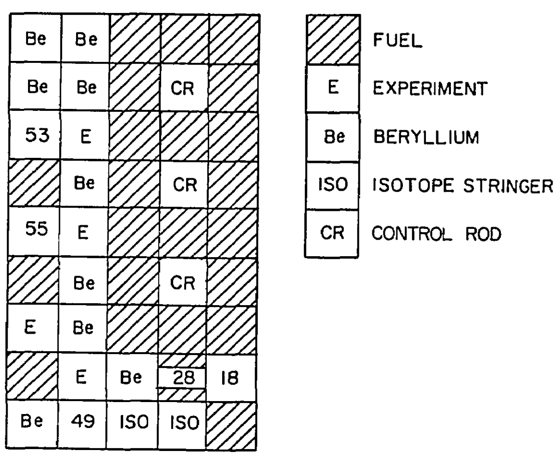


Fig. 7 - Schematic drawing of the Low Intensity Test Reactor core lattice showing the relative locations of the fuel and experimental positions C-18 and C-55, which contained dummy fuel elements suitable for the exposure of experimental irradiation assemblies

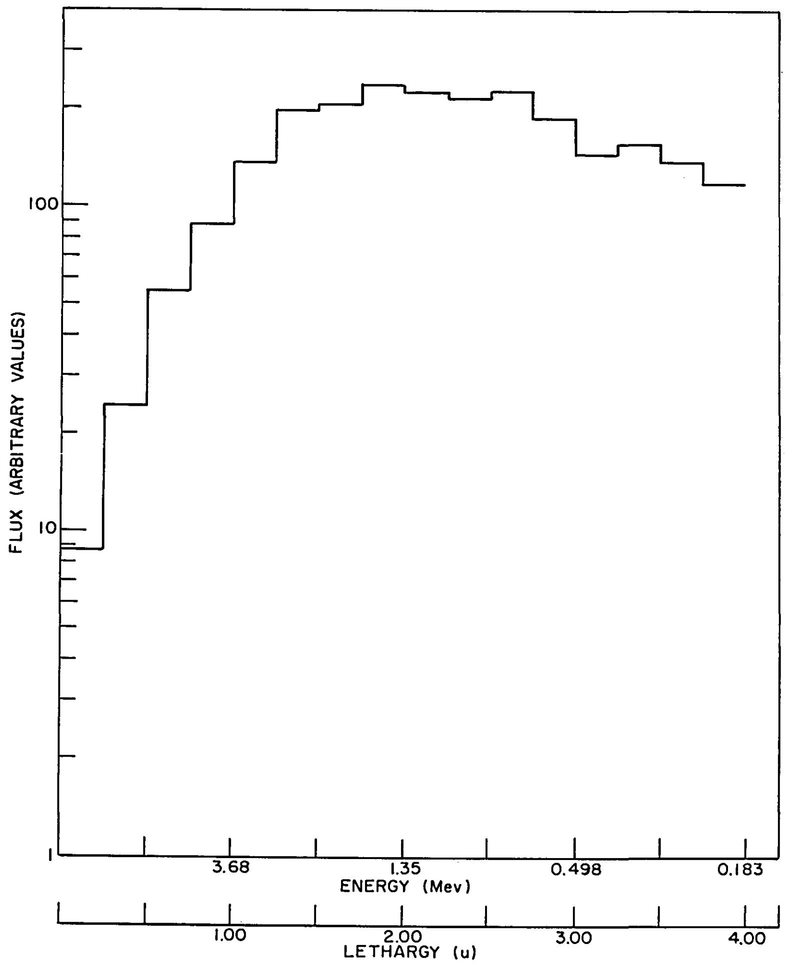


Fig. 8 - Graphical representation of the neutron spectrum in the C-18 core lattice position of the Low Intensity Test Reactor

the basis for all of the neutron dosimetry results in this report. The data points from these accelerator studies, as well as the curves through the points by Carroll and Smith (10), Helm (11), and Barrall and McElroy (12) are shown in Fig. 10. (The references noted on Fig. 10 are contained in Ref. 12.) Although the curves disagree significantly in the range between 6 and 14 Mev, there are few neutrons in a reactor spectrum in this range, so this disagreement will have little effect on the cross section. The really significant differences between the three curves are at the lower Mev portion of the curves, where the preponderance of activations occur. The effect of the curves in terms of activation cross sections for a fission spectrum is shown in Fig. 11. As can be seen, there are wide differences in the portion of a fission spectrum considered by the three curves, between 2 and 8 Mev, and particularly between 2 and 5 Mev. These differences give rise to the fission-averaged cross section for iron activation of 68 mb by Shure (13) from the work of Carroll and Smith (10), 81 mb by Helm (11), and 97.2 mb by Barrall and McElroy (12).

The use of the lowest cross section, 68 mb, will yield a fission-averaged flux 1-1/2 times that which would be determined by using the highest cross section, 97.2 mb. The

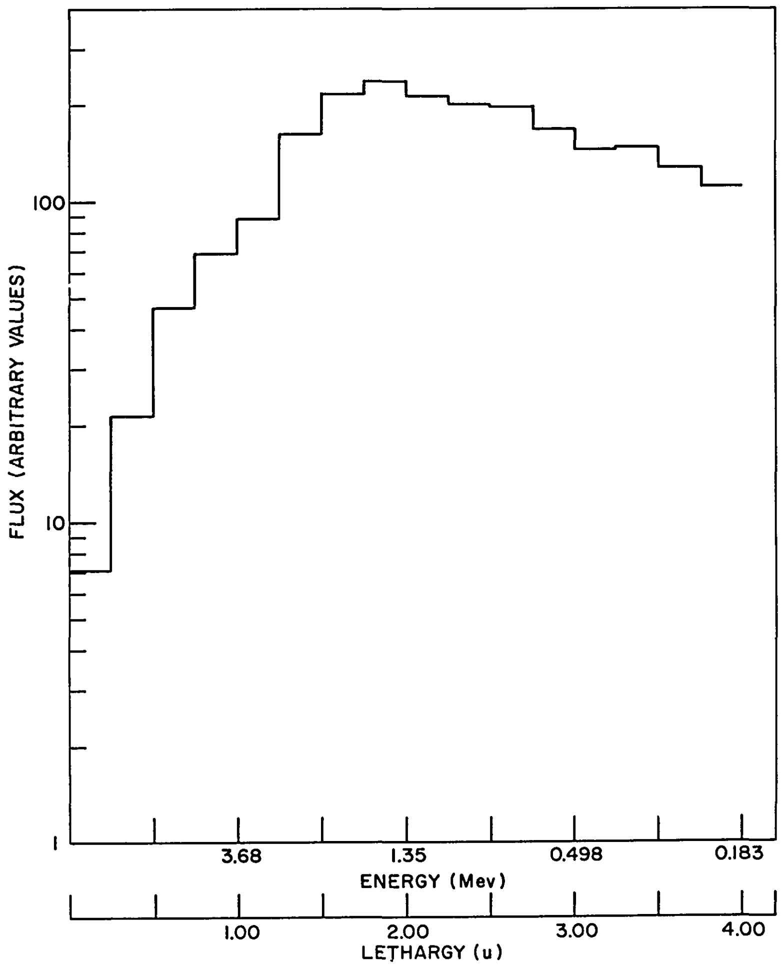


Fig. 9 - Graphical representation of the neutron spectrum in the C-55 core lattice position of the Low Intensity Test Reactor

implications of this kind of variation upon a meaningful analysis of the SM-1A pressure vessel are obvious. Fortunately, two reactors are involved in the SM-1A analysis. This fact may be the key to future analyses of radiation damage status of reactor pressure vessels. As will be shown in a subsequent section, the analysis may be performed by dividing the neutron flux per Mw-yr at the SM-1A vessel wall into the total exposure measured in the LITR which yields the maximum allowable NDT increase. Thus, the use of a cross section giving high flux values will yield higher values for both reactor systems. Similarly, the use of a cross section giving lower flux values will yield lower values for both reactor systems, and the resultant Mw-yr exposure limit will be virtually the same as that determined from the high cross section value.

Since the analysis relative to the SM-1A vessel condition is therefore relatively insensitive to the fission-averaged cross section for monitor activation, the selection of this constant from the three choices is not critical to a meaningful analysis. Based upon the correlation of vessel wall fluxes with SM-1A vessel mock-up fluxes (14), the Army has selected the 81-mb cross section of Helm to be used as the basis for the SM-1A vessel exposure lifetime analysis presented in this report.

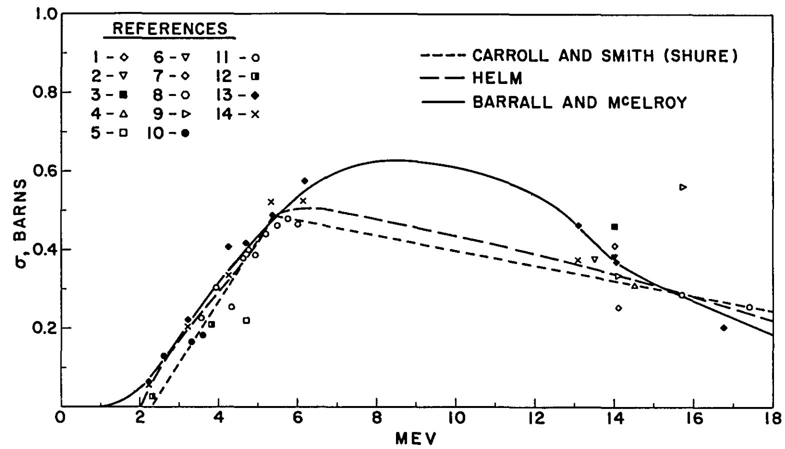


Fig. 10 - Cross-section measurements versus energy (Mev) of the $Fe^{54}(n,p)Mn^{54}$ reaction with the interpretation of the points by three investigators. Carroll and Smith obtained several of the data points and constructed the short-dashed-line curve. Shure apparently used the data of Carroll and Smith to construct group-averaged cross sections for this reaction. References to the 14 sets of data points are contained in Ref. 12. (Photo courtesy of R. E. Dahl, Battelle-Northwest Laboratories.)

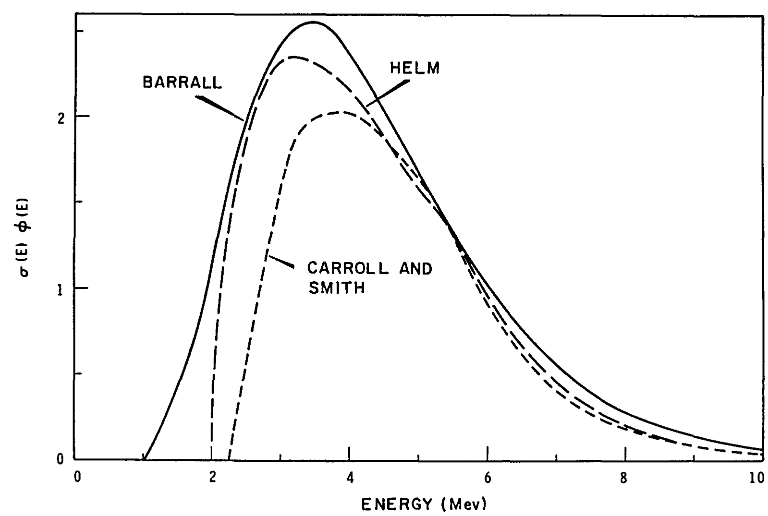


Fig. 11 - Response function of the $Fe^{54}(n,p)Mn^{54}$ reaction to the Watt fission spectrum as influenced by the interpretation of three investigators. The preponderance of activations for iron in a reactor neutron spectrum occur in the region of 2 to 5 Mev; thus the data point interpretations from Fig. 10 result in significant variations in the activation cross section for iron indicated in this figure. (Photo courtesy of R. E. Dahl, Battelle-Northwest Laboratories.)

## SM-1A PRESSURE VESSEL EXPOSURE LIMIT ANALYSIS

The neutron exposures shown in Table 1 which relate to the NDT increases of the vessel steel, and the neutron fluxes at the vessel wall monitor position shown in Fig. 2, were all determined from an iron-activation cross section of 68 mb, and reported as n/cm$^2$ or n/cm$^2$-sec $>$1 Mev, assuming a fission spectrum. Data presented in this report subsequent to Table 1 has been adjusted to Helm's 81-mb cross section and for comparison purposes to Barrall and McElroy's 92.1 and Shure's 68 as well. In order to convert the Table 1 data by any of the calculated cross-section values, the following sequence is employed. In the activation equation, 68 mb appears in the numerator and, since neutrons of energies $>$1 Mev comprise 69.3% of a fission spectrum, 0.693 appears in the denominator. It is convenient, then to perform the calculation

$$\frac{68 \text{ mb}}{0.693} = 98 \text{ mb} \tag{2}$$

and employ 98 mb for the effective cross section for neutrons $>$1 Mev in a fission spectrum as a starting point for the spectral influenced flux corrections.

The subsequent vessel exposure limit analysis presents a technique for the conversion of measured SM-1A vessel wall monitor position fluxes into spectrally corrected fluxes at the inner edge of the pressure vessel wall in terms of n/cm$^2$-sec above $E_L$ values of 1.0, 0.5, and 0.183 Mev. Similarly, the exposures measured in the LITR and SM-1A for the maximum NDT-increase data are converted into exposures in terms of n/cm$^2$ above the three $E_L$ values. The corrected vessel wall fluxes are then integrated in terms of exposures for one Mw-yr and are then divided into the exposure calculated for the maximum allowable ΔNDT to yield a maximum Mw-yr power output for the SM-1A reactor.

Contained within this analysis is the consideration of the damaging potential of the LITR reactor versus that of the SM-1A reactor. Calculations relating to this particular step are performed only for the 340°F ΔNDT value for the irradiated forging steel.

The spectral average cross sections determined by Battelle-Northwest for the $Fe^{54}$(n,p)$Mn^{54}$ reaction in the SM-1A and LITR are given in Table 3. The cross sections for activation above $E_L$ values of 1.0, 0.5, and 0.183 Mev are shown, as determined by using the fission-averaged cross sections of Shure, Helm, and Barrall and McElroy. The calculational sequence presented is performed only for the Helm cross-section value, though use of the other cross-section values for adjustment of both reactor spectra would lead to essentially the same ultimate conclusions.

### Vessel Exposure Limit Calculations

1. Adjustment of the SM-1A Monitor Position (MP) flux to the calculated spectrum:

| $E_L$ (Mev) | Measured Flux $\times \dfrac{\bar{\sigma}\ 98 \text{ mb (fission spectrum) (Eq. (2))}}{\bar{\sigma} \text{ Helm (calculated spectrum)}}$ |
|---|---|
| $>$1.0 | $1.11 \times 10^{11} \times \dfrac{98 \text{ mb}}{126 \text{ mb}} = 8.63 \times 10^{10}$ n/cm$^2$-sec |
| $>$0.5 | $1.11 \times 10^{11} \times \dfrac{98 \text{ mb}}{78.9 \text{ mb}} = 1.38 \times 10^{11}$ n/cm$^2$-sec |
| $>$0.183 | $1.11 \times 10^{10} \times \dfrac{98 \text{ mb}}{56.7 \text{ mb}} = 1.92 \times 10^{11}$ n/cm$^2$-sec |

(3)

Table 3
Spectral Averaged Cross Sections for the $Fe^{54}$(n,p) Reaction for Irradiation Positions in the LITR and SM-1A Reactors

| | $\sigma_{Fe}$ (mb) | | | | | | | | | Flux $\phi$ Relative to $\phi(E>1)$ at Pressure Vessel | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $\phi$ > 1 Mev | | | $\phi$ > 0.5 Mev | | | $\phi$ > 0.18 Mev | | | | | |
| | Shure | Helm | Barrall | Shure | Helm | Barrall | Shure | Helm | Barrall | $\phi(E>1)$ | $\phi(E>0.5)$ | $\phi(E>0.18)$ |
| LITR: | | | | | | | | | | | | |
| C-18 | 79.3 | 99.6 | 118.9 | 54.3 | 68.2 | 81.4 | 41.9 | 52.7 | 62.9 | N.A. | N.A. | N.A. |
| C-55 | 70.9 | 89.0 | 108.6 | 48.4 | 60.8 | 74.2 | 36.9 | 46.3 | 56.5 | N.A. | N.A. | N.A. |
| SM-1A: | | | | | | | | | | | | |
| (Above-Core) | 142.4 | 167.9 | 196.0 | 121.6 | 143.5 | 167.4 | 111.4 | 131.4 | 153.3 | N.A. | N.A. | N.A. |
| (At Monitor) | 105.3 | 126.0 | 148.5 | 65.9 | 78.9 | 93.0 | 47.5 | 56.7 | 66.9 | 1.16 | 1.88 | 2.61 |
| (At Pressure Vessel) | 105.2 | 125.8 | 148.5 | 64.0 | 76.6 | 90.4 | 45.6 | 54.6 | 64.4 | 1.0 | 1.64 | 2.31 |

2. Extrapolation of the SM-1A MP flux to the SM-1A Pressure Vessel Wall (PVW) flux:

| $E_L$ (Mev) | Adjusted SM-1A MP Flux $\times \frac{\text{PVW relative flux (Table 3)}}{\text{MP relative flux}}$ = PVW Flux | |
|---|---|---|
| | | (4) |
| >1.0 | $8.63 \times 10^{10} \times \frac{1.0}{1.16} = 7.41 \times 10^{10}$ n/cm$^2$-sec | |
| >0.5 | $1.38 \times 10^{11} \times \frac{1.64}{1.88} = 1.21 \times 10^{11}$ n/cm$^2$-sec | |
| >0.183 | $1.92 \times 10^{11} \times \frac{2.31}{2.61} = 1.69 \times 10^{11}$ n/cm$^2$-sec | |

3. Convert the SM-1A PVW flux (measured at 8.65 Mw) to exposure per Mw-yr SM-1A PVW flux:

| $E_L$ (Mev) | $\frac{\text{n/cm}^2\text{-sec at 8.65 Mw}}{\text{8.65 Mw}} \times$ sec/yr = n/cm$^2$ for 1 Mw-yr | |
|---|---|---|
| | | (5) |
| >1.0 | $\frac{7.41 \times 10^{10}}{8.65} \times 3.154 \times 10^7 = 2.70 \times 10^{17}$ | |
| >0.5 | $\frac{1.21 \times 10^{11}}{8.65} \times 3.154 \times 10^7 = 4.41 \times 10^{17}$ | |
| >0.183 | $\frac{1.69 \times 10^{11}}{8.65} \times 3.154 \times 10^7 = 6.16 \times 10^{17}$ | |

4. Adjustment of the LITR C-18 neutron exposure for the forging ΔNDT of 340°F, to the calculated spectrum for that position:

| $E_L$ (Mev) | C-18 Exposure (Fission Spectrum) $\times \frac{\bar{\sigma}\text{98 mb (Eq. (2))}}{\bar{\sigma}\text{ Helm (calc. spectrum)}}$ = C-18 Exposure (Calc. Spectrum) | |
|---|---|---|
| | | (6) |
| >1.0 | $2 \times 10^{19} \times \frac{98\text{ mb}}{99.6\text{ mb}} = 1.97 \times 10^{19}$ | |
| >0.5 | $2 \times 10^{19} \times \frac{98\text{ mb}}{68.2\text{ mb}} = 2.87 \times 10^{19}$ | |
| >0.183 | $2 \times 10^{19} \times \frac{98\text{ mb}}{52.7\text{ mb}} = 3.72 \times 10^{19}$ | |

5. Calculation of the damage index for the LITR C-18 position spectrum relative to the damage index for the SM-1A PVW spectrum:

| $E_L$ (Mev) | Damage Equivalence Factor = $\dfrac{K_{(\text{LITR C-18})}\ (\text{Eq. (1)})}{K_{(\text{SM-1A PVW})}\ (\text{Eq. (1)})}$ |
|---|---|
| $>1.0$ | 0.914 |
| $>0.5$ | 1.028 |
| $>0.183$ | 1.116 |

(7)

6. Adjustment of the LITR C-18 exposures to the SM-1A PVW spectrum in terms of damage equivalence:

| $E_L$ (Mev) | LITR C-18 Exposure × Damage Equivalence = LITR C-18 Exposure Adjusted to SM-1A PVW |
|---|---|
| $>1.0$ | $1.97 \times 10^{19} \times 0.914 = 1.80 \times 10^{19}$ |
| $>0.5$ | $2.87 \times 10^{19} \times 1.028 = 2.95 \times 10^{19}$ |
| $>0.183$ | $3.72 \times 10^{19} \times 1.116 = 4.15 \times 10^{19}$ |

(8)

7. Calculation of Mw-yr for the SM-1A PVW to reach 340°F ΔNDT from LITR C-18 forging data:

| $E_L$ (Mev) | $\dfrac{\text{Adjusted LITR C-18 Exposure Yielding 340°F ΔNDT (Eq. (8))}}{\text{One-Mw-yr Exposure at SM-1A PVW (Eq. (5))}}$ = Reactor power output to reach critical NDT on SM-1A PVW |
|---|---|
| $>1.0$ | $\dfrac{1.80 \times 10^{19}\ \text{n/cm}^2}{2.70 \times 10^{17}\ \text{n/cm}^2\text{-Mw-yr}} = 66.7$ Mw-yr |
| $>0.5$ | $\dfrac{2.95 \times 10^{19}\ \text{n/cm}^2}{4.41 \times 10^{17}\ \text{n/cm}^2\text{-Mw-yr}} = 66.9$ Mw-yr |
| $>0.183$ | $\dfrac{4.15 \times 10^{19}\ \text{n/cm}^2}{6.16 \times 10^{17}\ \text{n/cm}^2\text{-Mw-yr}} = 67.4$ Mw-yr |

(9)

The Mw-yr output values calculated under Eq. (9) are effectively all 67 Mw-yr. The slight variations result from the accumulated rounding of different values during the analysis.

The use of dislocation production as a damage index (Eq. (7)) for adjusting the relative damaging potential of incident neutron spectra is validated by the results shown in item 7 (Eq. (9)) above. If the damage equivalence factor is not employed, the critical SM-1A reactor power output would be calculated as follows:

| $E_L$ (Mev) | LITR C-18 Exposure Adjusted to C-18 Spectrum for 340°F ΔNDT (Eq. (6)) / SM-1A PVW Exposure for 1 Mw-yr Adjusted to SM-1A PVW Spectrum (Eq. (5)) = SM-1A power output not adjusted for damage index |
|---|---|
| $>1.0$ | $\frac{1.97 \times 10^{19}}{2.70 \times 10^{17}} = 73.0$ Mw-yr |
| $>0.5$ | $\frac{2.97 \times 10^{19}}{4.41 \times 10^{17}} = 65.1$ Mw-yr |
| $>0.183$ | $\frac{3.72 \times 10^{19}}{6.16 \times 10^{17}} = 60.4$ Mw-yr |

(10)

A comparison of the critical power output values calculated by the method of Eq. (9) versus that of Eq. (10) reveals that the values for neutrons above lower energy limit 0.5 Mev are very similar, while the critical power output values for neutrons above lower energy limits 1.0 and 0.183 Mev are different. This further substantiates the thesis that a neutron exposure criterion of n/cm$^2$ $> E_L$ 0.5 Mev yields the most accurate neutron dosimetry values when comparing results from significantly differing reactor neutron energy spectra. However, it should be noted that use of the 1-Mev lower energy limit without application of the damage-index correction yields an exposure limit of 73 Mw-yr, versus 66.9 Mw-yr for the damage-index corrected value. Thus, in this case, if the adjusted values are assumed to be absolute, the error from using an unadjusted value, but based on a calculated spectrum, is only about 10%. This is not a serious variation when all of the other potential errors in such analyses of radiation damage are considered, but may be critical where the possible safety of an operating reactor is concerned.

The spectrally adjusted neutron exposure for NDT increase of the A350-LF1 (Modified) plate and forging shown in Table 1 have been plotted in Fig. 12. In this figure, the 340°F ΔNDT forging data (as well as the 330°F ΔNDT plate value from the same experiment in the C-18 lattice position) have been additionally corrected for the damage index (Eq. (7)). Since the other data points do not enter into the critical exposure limit analysis, this step was neglected in the determination of those adjusted exposures. Figure 12 also indicates the maximum permissible pressure vessel wall neutron exposures for neutrons above lower energy limits 1.0, 0.5, and 0.183 Mev as calculated in Eq. (9).

By applying the damage-index adjusted values of Eq. (8), it is possible to estimate the remaining time (in terms of reactor power output-Mw-yr) for the SM-1A reactor vessel to reach the critical exposure limit. The SM-1A reactor has operated for 26.6 Mw-yr through the end of Core II, October 1965. Starting with Core III and using the $E_L$ 0.5-Mev value (Eq. (9)), the SM-1A reactor still should have

$$67 \text{ Mw-yr} - 26.6 \text{ Mw-yr} = 40.4 \text{ Mw-yr}$$

remaining before reaching the critical ΔNDT value of 340°F.

## SUMMARY

The problem of assessing the critical neutron exposure responsible for increasing the NDT of the SM-1A reactor pressure vessel to an untenable level could not be treated

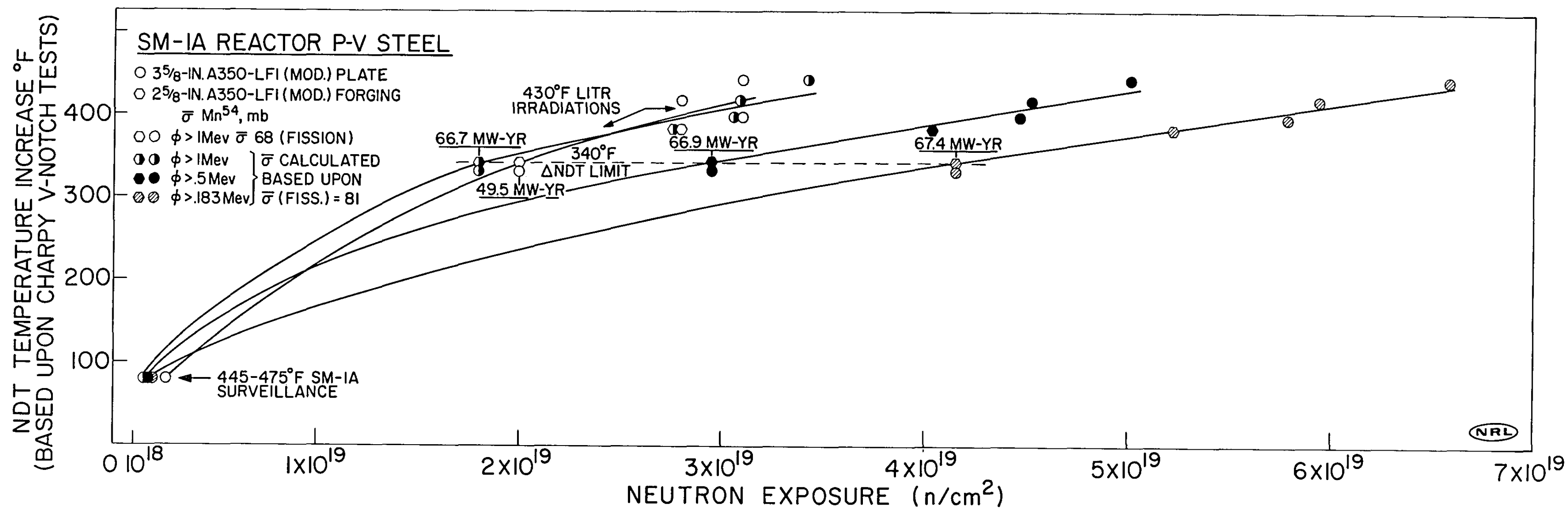


Fig. 12 - Increase in the NDT temperature of SM-1A reactor pressure vessel steel versus neutron exposures on a linear scale, in terms on $n/cm^2$ required to produce the critical power output level of ΔNDT 340°F. The neutron exposures have been converted from a fission-spectrum assumption basis to account for the calculated spectra of the SM-1A reactor and the LITR using a fission-averaged cross section of 81 mb for the $Fe^{54}(n,p)Mn^{54}$ reaction. The critical power output limit for the SM-1A reactor will be effectively 67 Mw-yr, regardless of whether neutrons >1, >0.5, or >0.183 Mev in the calculated spectra are reported.

in the manner as with larger nuclear plants, that is, by radiation damage surveillance at the pressure vessel wall. The next best alternative involved the exposure of representative specimens of the vessel steel in a test reactor (the LITR) under temperature and exposure conditions which simulated the SM-1A operating conditions, as determined by neutron dosimeters located at the pressure vessel wall and by records of the operating temperature.

In order to project a critical exposure limit based upon data from experiments conducted in the LITR and neutron dosimetry data from the SM-1A reactor, a multistep procedure was necessary. This procedure involved: (a) adjustment of the measured SM-1A neutron flux, based on dosimeters, by the calculated neutron energy spectrum at the vessel wall, (b) extrapolation of the adjusted dosimeter position flux to the SM-1A pressure vessel wall (inside the cladding overlay), (c) conversion of the SM-1A pressure vessel flux to an exposure per megawatt-year for the pressure vessel wall location, (d) adjustment of the LITR neutron exposure required to produce an NDT increase of 340°F for the SM-1A steel to the calculated spectrum for the LITR position, (e) determination of a damage factor, a revised cross section based upon atomic-displacement production in the specific spectrum, and correlation of these factors for the two spectra involved, (f) adjustment of the LITR exposure values to the SM-1A pressure vessel spectrum in terms of damage equivalence, and (g) extrapolation of effective damaging exposure per megawatt-year to provide a total reactor power output (accumulated Mw-yr) which yields the exposure that will produce the maximum allowable vessel ΔNDT (340°F) for safe operation.

Through this procedure, it has been possible, using experimental NDT data determined from experiments in the LITR and neutron dosimetry values from the SM-1A, to adjust the data for the most realistic projection of the reactor power output required to reach a predetermined maximum allowable ΔNDT for the SM-1A reactor vessel. The projected maximum permissible reactor output is 67 Mw-yr. This analysis was made possible by advances in the neutron physics and computer techniques for calculating neutron spectra in nuclear reactor core areas, as well as by improved cross sections for the key activation dosimetry reaction $Fe^{54}(n,p)Mn^{54}$. Furthermore, these tools permit a sophisticated evaluation of radiation damage through the application of a damage model such as that of total-displacement production as used in this report. Also, by use of these advanced techniques, it is possible to assess radiation damage in terms of neutron exposures above various lower energy limits for a specific neutron spectrum. In this analysis, three energy limits have been used, the conventional 1 Mev, the preferred limit of 0.5 as defined by Dahl and Yoshikawa (5), and 0.183 as a selected minimum exposure cut-off.

This pioneering analysis, while considered a great improvement over the practice of assuming a fission spectrum and extrapolating neutron exposure above 1 Mev, is subject to certain limitations. These relate to possible errors in the dosimetry analysis (the measurement of flux values), in the calculation of neutron spectra, in the application of a damage model, and in the application of limited experimental data on NDT increases for this particular steel. Possible errors associated with flux calculation have been estimated to be about ± 5% by the Idaho Nuclear Corporation, which performed these analyses. Regardless of whether this error was plus 5% or minus 5%, the error can be considered to be constant, since the counting equipment and techniques have been maintained essentially unchanged during the period of time in which the analyses presented in this report were performed. Thus, it is considered reasonable to state that the error in counting and in the determination of this fission-average neutron flux is negligible. In the case of the calculated spectra, however, consideration of possible errors may be more realistic since the calculation involved two reactors in which the materials between the core and the point of spectrum determination were physically quite different. No attempt has been made to estimate this error. It is pointed out, however, that the same library of physical constants was employed for all the calculations, and the values utilized were the very latest available. In the case of the damage index, it is not possible to estimate variations which might apply if a different model were used; however, it is considered significant that, by applying the damage-index exposure values using the three exposure levels, lower energy limits, 1.0, 0.5, and 0.183, the same extrapolated reactor power output is obtained for reaching a given NDT on the vessel. Thus, this technique provides for a significant improvement over the direct extrapolation of results as in Eq. (8) above. In regard to the experimental data on NDT, no error analysis has been made; however, the relative uniformity of these data especially when corrected for spectrum variations are considered excellent.

In summary, it may be stated that, by adjusting data for specific spectra in the two reactors, it has been possible to establish more accurately an anticipated maximum allowable exposure in terms of reactor power output for the development of a critical NDT value for the SM-1A reactor pressure vessel.

## CONCLUSIONS

A critical exposure limit has been calculated for the SM-1A reactor pressure vessel from data obtained both in the SM-1A and in the Low Intensity Test Reactor. By considering the relative damaging potential of the neutron spectra in both of these reactors, it has been shown, accounting for neutrons of energies $>0.5$ Mev, that a 67-Mw-yr operating output can be projected before the neutron-induced NDT increase for the SM-1A pressure vessel reaches a critical point. If the reactor spectra were assumed to be fission spectra however, the Mw-yr critical power output limit for neutrons $>1$ Mev, for instance, would be 49.5. Thus, by considering the neutrons to be actually present in a reactor spectrum rather than by assuming a fixed distribution, a more precise representation of the neutron exposure required to effect an NDT increase can be determined, and less conservatism need be applied in projecting safe operating limits.

In a more general context, it is believed that this analysis demonstrates that data on radiation-induced embrittlement of steels from test reactor irradiations may be meaninfully applied to operating power reactor pressure vessels if the damage-causing neutron exposures in the test reactor are adjusted for the neutron spectrum and then compared with similarly adjusted values for the operating reactor vessel. Furthermore, it is believed that the $\Delta$NDT data may be further refined and made more useful through the correlation of the relative damaging potential of two different reactor spectra by applying a damage-index adjustment of exposure values. As described above, it appears that the application of neutron exposure data above a lower energy level of 0.5 Mev with a calculated spectrum provides the best correlation criterion for mechanical-property data when two different reactor environments are involved. That is, independent of any damage-index adjustment, this exposure limit between two different reactor spectra provides the best means for direct comparison.

It is felt that this procedure will have many valuable applications in the consideration of the potential hazards associated with radiation embrittlement of other reactor pressure vessels.

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Nuclear Power Field Office of the U.S. Army Corps of Engineers to whom special thanks must go for their permission to publish this first analysis of neutron spectral considerations in the evaluation of a critical NDT condition for an operating power reactor vessel.

## REFERENCES


1. Pellini, W.S., and Puzak, P.P., "Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures," NRL Report 5920, Mar. 15, 1963
2. Steele, L.E., and Hawthorne, J.R., "Radiation Damage Surveillance of the SM-1A Reactor Pressure Vessel," NRL Memorandum Report 1402, Feb. 1963
3. Steele, L.E., et al., "Irradiated Materials Evaluation and Reactor Pressure Vessel Surveillance for the Army Nuclear Power Program," NRL Memorandum Report 1644, Sept. 1, 1965
4. Dahl, R.E., "Measuring and Correlating Neutron Exposure and Damage in Graphite," HW-79793, Hanford Atomic Products Operation, General Electric Co., Dec. 1963
5. Dahl, R.E., and Yoshikawa, H.H., "Neutron Exposure Correlation for Radiation-Damage Studies," Nuclear Science and Engineering 21:312-318 (1965)
6. Duane, B.H., "Neutron and Photon Transport Plane-Cylinder-Sphere (GE-ANPD) Program S Variational Optimum Formulation," General Electric Company, XOC-9-118 (1959)
7. Bengston, J., et al., "2DXY Two Dimensional, Cartesian Coordinate Sn Transport Calculation," USAEC Research and Development Report AGNTM-392, June 1961
8. Kinchin, G.H., and Pease, R.S., "The Displacement of Atoms in Solids by Radiation," Reports on Progress in Physics (Phys. Soc., London) 18:1-15 (1955)
9. Rossin, A.D., "Degradation of Impact Energy of Steel as a Function of Neutron Exposure," Trans. Amer. Nucl. Soc. 6(2):389 (Nov. 1963)
10. Carroll, Edward E., Jr., and Smith, George G., "Iron-54 (n,p) Cross-Section Measurement," Nuclear Science and Engineering, 22:411-415 (1965)
11. Helm, J.W., "High-Temperature Graphite Irradiations: 800 to 1200 Degrees C: Interim Report No. 1," BNWL-112, Battelle-Northwest Laboratories, Sept. 1965
12. Barrall, R.C., and McElroy, W.N., "Neutron Flux Spectra Determination by Foil Activation," AFWL-TR-65-34 (Vol. II), Air Force Weapons Laboratory, Aug. 1965
13. Shure, K., "Radiation Damage Exposure and Embrittlement of Reactor Pressure Vessels," WAPD-TM-471, Westinghouse Atomic Power Department, Nov. 1964
14. Moote, F.G., et al., "Fast Neutron Flux in a Mockup of SM-1A Core and Vessel," ACNP-63586, Allis-Chalmers Manufacturing Co., Atomic Energy Division, Aug. 1963

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13. ABSTRACT

The pressure vessel of the Army SM-1A reactor is located close to the active core in such a manner that the neutron exposure is relatively high; consequently, the pressure vessel steel undergoes a relatively rapid rise in the ductile-brittle transition temperature. The maximum permissible $\Delta$NDT for the SM-1A is established by the Army as 340°F. Since it is physically impossible to irradiate surveillance test specimens at the SM-1A vessel wall, only the neutron flux was measured at the wall, and representative test specimens were irradiated in a test reactor, the Low Intensity Test Reactor (LITR). In translating the $\Delta$NDT versus neutron exposure data from the LITR to the case of the SM-1A reactor vessel wall, the neutron spectra of the two reactors were used to adjust both the SM-1A reactor vessel flux and the LITR exposure values in terms of $n/cm^2 > 1.0$, 0.5, and 0.183 Mev. Since the distribution of neutrons by energy groups was different within each reactor at the specific location of interest, that is, the vessel wall of the SM-1A and an in-core location of the LITR, the damaging potential of the SM-1A reactor spectrum location was related to that of the LITR.

With damage equivalence established between the two reactors, a critical neutron exposure ($n/cm^2 > 0.5$ Mev) may be projected for producing the maximum $\Delta$NDT on the SM-1A reactor vessel wall. By relating this critical exposure to SM-1A reactor operations, a critical power output level of 67 Mw-yr was established. The same Mw-yr critical output levels will be obtained if the neutron exposure is reported as $n/cm^2 > 1$ Mev or $n/cm^2 > 0.183$ Mev. If the reactor spectra were assumed to be fission spectra, however, the Mw-yr critical power output limit for neutrons > 1 Mev, for instance, would be 49.5. Thus, by considering all of the neutrons actually present in a reactor spectrum rather than by assuming a fixed distribution, a more precise calculation of the neutron exposure required to effect a NDT increase can be determined, and less conservatism need be applied in projecting safe operating limits.

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